NON-PUBLIC?: N
ACCESSION #: 9201210249
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Nine Point Unit 2 PAGE: 1 OF 9

DOCKET NUMBER: 05000410

TITLE: Reactor Scram on Low Reactor Water Level due to Loss of Feedwater Pumps
EVENT DATE: 12/12/91 LER #: 91-023-00 REPORT DATE: 01/13/92

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 055

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: Mr. Alan DeGarcia, Manager TELEPHONE: (315) 349-7531
Operations NMP2

COMPONENT FAILURE DESCRIPTION:
CAUSE: SYSTEM: COMPONENT: MANUFACTURER:
REPORTABLE NPRDS:

SUPPLEMENTAL REPORT EXPECTED: No

ABSTRACT:

On December 12, 1991, at 0322 hours, with the reactor mode switch in the "RUN" position and the plant operating at approximately 55 percent rated thermal power, Nine Mile Point Unit 2 (NMP2) experienced a reactor scram on a reactor vessel low water level (Level 3) signal. Specifically, following the startup of Feedwater System pump 2FWS-P1A in support of raising plant power, a Condensate System (CNM) and Feedwater System (FWS) transient occurred, resulting in the loss of both FWS pumps. Reactor vessel water level lowered to 159.3 inches (Level 3 trip setpoint), initiating an automatic reactor scram signal. On December 12, 1991, at 1023 hours, while operators were attempting to return the Reactor Water Cleanup (WCS) System to an operable status following the scram, WCS isolated on a high differential flow signal (Engineered Safety Feature actuation).

The root cause for the reactor scram was determined to be poor work practices. The preliminary cause for the WCS isolation is inadequate system design.

Immediate actions included restoring reactor vessel inventory and commencing a controlled plant shutdown. Additional corrective actions include: 1) administering disciplinary action for individuals involved in the scram; 2) evaluating and revising Operating Procedure N2-OP-3, "Condensate and Feedwater System"; 3) troubleshooting electrical circuitry for Condensate pump 2CNM-P1C; and 4) completing the WCS isolation root cause analysis.

END OF ABSTRACT

I. DESCRIPTION OF EVENT

On December 12, 1991, at 0322 hours, with the reactor mode switch in the "RUN" position and the plant operating at approximately 55 percent rated thermal power, Nine Mile Point Unit 2 (NMP2) experienced a reactor scram on a reactor vessel low water level (Level 3) signal. Specifically, following the startup of Feedwater System pump 2FWS-P1A, in support of raising plant power, a Condensate System (CNM) and Feedwater System (FWS) transient occurred which resulted in the loss of Feedwater System flow. Reactor pressure vessel (RPV) water level lowered to 159.3 inches (Reactor Protection System Level 3 trip setpoint), initiating an automatic reactor scram signal.

Prior to starting Feedwater pump 2FWS-P1A, the Condensate and Feedwater pumps were operating in a 2-2-1 alignment (2 Condensate pumps, 2 Condensate Booster pumps and 1 Feedwater pump). The control switches for the third Condensate pump and the third Condensate Booster pump were in the pull-to-lock position.

When the second Feedwater pump (1A) was started, the Condensate System Flow, as measured on the suction side of the Condensate Booster pumps, began to increase rapidly. Recognizing lowering pump suction pressures, operators attempted to manually start Condensate pump 2CNM-Pl C from switchgear 2NNS-SWG011. It failed to start. The flow reached 25,190 gallons per minute (GPM) and was still increasing when the "B" Condensate Booster pump tripped on low suction pressure. The third Condensate Booster pump ("C") started when Control Room operators took the control switch out of the pull-to-lock position. The "A" Condensate Booster pump then tripped on low suction pressure. RPV water level decreased to Level 4 (178.3 inches). Operators then stopped 2FWS-P1A manually and the

Reactor Recirculation System ran back flow automatically. When operators restarted the "A" Condensate Booster pump, the "B" Feedwater pump tripped. With the loss of both running Feedwater pumps, RPV water level reached Level 3 (159.3 inches), initiating an automatic reactor scram signal. RPV water level continued to lower until Level 2 (108.8 inches) was reached, at which point High Pressure Core Spray (HPCS) and Reactor Core Isolation Cooling (RCIC) automatically initiated and began injecting into the RPV. These systems recovered RPV level to the normal operating band at which time they were secured. Feedwater pump 2FWS-P1A was successfully restarted and used to control RPV water level during the ensuing plant shutdown.

As a result of RPV water level reaching Level 3, the following automatic actuations occurred:

o Primary Containment isolation Group 4 isolated

o Reactor scram

I. DESCRIPTION OF EVENT (cont.)

As a result of RPV water Level 2 being reached, the following automatic actuations occurred:

o Primary Containment isolation Groups 2, 3, 6, 7, 8 and 9 isolated

o The Reactor Building isolated and the Standby Gas Treatment System (GTS) initiated

o Control Building Special Filter Train initiated

o Reactor Recirculation System pumps auto tripped

o HPCS and RCIC initiated, and

o The Division Ill Emergency Diesel Generator auto started

All automatic responses for RPV water Levels 2, 3, and 4 were verified to have properly occurred.

An Unusual Event emergency classification was declared at 0332 hours as a result of the Emergency Core Cooling System (ECCS) injection into the RPV on a valid initiation signal. The Unusual Event was terminated at 0437 hours.

At approximately 1023 hours on December 12, 1991, while operators were attempting to return the Reactor Water Cleanup System (WCS) to an operable status following the reactor scram (0322 hours), WCS isolated on a high differential (delta) flow signal. WCS is designed to isolate when a high system differential flow condition is sensed following a 45 second time delay.

During the system fill and vent activities, the high delta flow timers initiated. Operators immediately tripped the inservice Reactor Water Cleanup pump 2WCS-P1B, and manually initiated the closure of the inboard and outboard Containment isolation valves (2WCS*MOV102 and 2WCS*MOV112 respectively) in an attempt to preclude this Engineered Safety Feature (ESF) actuation. However, the 45 second time delay expired and the ESF isolation signal was generated.

I. DESCRIPTION OF EVENT (cont.)

SCRAM EVENTS

TIME (12/12/91) ACTIVITY

03:20 2FWS-P1B, "B" Feedwater pump running
2CNM-P1A, "A" Condensate pump running
2CNM-P1B, "B' Condensate pump running
2CNM-P2A, "A" Condensate Booster pump running
2CNM-P2B, "B" Condensate Booster pump running
03:21:02 2FWS-P1A, "A" Feedwater pump started
03:21 Attempted manual start of Condensate pump
2CNM-P1C. Pump failed to start.
03:21:44 2CNM-P2B, "B" Condensate Booster pump tripped on
low suction pressure
03:21:47 2CNM-P2C, "C" Condensate Booster pump started
03:21:52 2CNM-P2A, "A" Condensate Booster pump tripped on
low suction pressure
03:21:59 2FWS-P1A, "A" Feedwater pump manually tripped
03:22:02 2CNM-P2A, "A" Condensate Booster pump restarted
03:22:04 2FWS-P1B, "B" Feedwater pump tripped on low
suction pressure
03:22:15 Reactor scram (RPV water Level 3 signal - 159.3
inches)
EOPs entered
Primary Con

ainment (Group 4) isolation
03:22:33 RPV water Level 2 signal (108.8 inches)
Primary Containment (Groups 3, 8 and 9)
isolation HPCS auto initiated (Division Ill
Emergency Diesel start)
03:22:38 Primary Containment (Groups 2, 6 and 7)
isolation
03:22:46 RCIC auto initiated
Main Turbine Generator tripped
03:23:13 HPCS reset
03:27:09 Secured RCIC injection (recirc mode)
03:32 Unusual Event declared (ECCS injection)

I. DESCRIPTION OF EVENT (cont.)

TIME (12/12/91) ACTIVITY

03:33 Reset scram
03:49 Inboard Mainsteam Isolation Valves closed
(manually)
04:11 Secured Division Ill Emergency Diesel Generator
04:18 RPV Water Level 8 (202.3 inches) trip signal,
2FWS-P1A, "A" Feedwater pump tripped
04:27 Residual Heat Removal System (RHS) Loop "A" in
reactor steam condensing mode, RHS Loop "B" in
Suppression Pool Cooling lineup
04:58 Exited EOPs
05:13 Main Stack Gaseous Effluent Monitoring System
(GEMS) declared inoperable
10:23 WCS isolation on Hi Delta Flow
11:17 Reset WCS isolation

II. CAUSE OF EVENT

A root cause investigation for the scram was performed utilizing Nuclear Division Procedure NDP-16.01, "Root Cause Evaluation."

The cause for the loss of Feedwater Flow was poor work practices. The Assistant Station Shift Supervisor (ASSS) involved failed to properly evaluate system conditions in making the decision not to start an additional Condensate and Condensate Booster pump prior to starting Feedwater pump 2FWS-P1A. Additional problems associated with this evolution were a breakdown in communications and failure of the chain of command. The Station Shift Supervisor (SSS) early in the shift (prior to scram) had directed the Chief Shift Operator (CSO) to complete preparation for startup of the third Condensate and second Feedwater pump. The ASSS and Reactor Operator performing the evolution did not receive this information, nor did the SSS provide oversight of this evolution. The SSS and CSO were not directly involved in the evolution nor consulted regarding use of the third Condensate and Condensate Booster pumps prior to the Feedwater pump startup.

A contributing factor was procedural inadequacy. The procedure being used, while it did provide for the option to start the additional pumps, did not provide detailed criteria to assist in making that decision. This led to exceeding the capacity of the two operating Condensate pumps during startup of a second Feedwater pump.

II. CAUSE OF EVENT (cont.)

A formal root cause investigation is presently being performed for the Reactor Water Cleanup System isolation. Preliminary conclusions have determined the most probable cause for this isolation was attributable to system design.

In support of WCS restart following the reactor scram, venting activities were performed to assure the system was filled. Minimum system cooldown (post-scram) resulted in high temperature and pressure in the isolated system. During venting (steam being released), system pressure lowered causing "flashing" and tripping of the high differential flow transmitters. Control Room operators responded to the 45 second timer initiation by securing the WCS pump and closing the system isolation valves. Valves were in the process of closing when the time delay signal was taken.

The type of transmitter used for WCS flow indication has a very fast response time and high sensitivity level. This has resulted in difficulty and potential risk of system isolation while attempting to maintain control of WCS flow during system manipulations. System manipulations often result in pressure transients, which can be misread by the differential flow logic as line breaks. Transients which cannot be corrected before the isolation signal completes its countdown (45 seconds), result in system isolations.

III. ANALYSIS OF EVENT

These events are reportable in accordance with 10CFR50.73 (a)(2)(iv), "Any event or condition that results in manual or automatic actuation of an Engineered Safety Feature (ESF), including the Reactor Protection System (RPS)." The reactor scram and the various safety system actuations including: the Division III ECCS actuation (covering the Division III Emergency Diesel Generator start); Secondary Containment isolation and automatic start of the Standby Gas Treatment System; the Control Room Special Filter Train start; the WCS isolation (Groups 6 and 7 containment isolation); and remaining containment isolations (Groups 2, 3, 4, 8 and 9) were automatic ESF actuations.

The reactor scram that resulted from a Level 3 reactor low water level trip signal occurred due to a loss of Feedwater System flow. The spectrum of events (including the Division III ECCS and all ESF actuations discussed above) that occurred as a result of the loss of Feedwater flow are bounded within the analysis of the "Loss of Feedwater Flow" event discussed in the USAR Section 15.2.7.

III. ANALYSIS OF EVENT (cont.)

The automatic actuations of the Reactor Core Isolation Cooling System and the High Pressure Core Spray System, with a subsequent coolant injection, was a conservative plant response with minimal plant impact and no resultant impact on public safety. HPCS and RCIC are designed to restore reactor water level to normal operating band to maintain fuel integrity. The lowest water level attained during this event was 106 inches. Reactor water level was restored to normal approximately 35 seconds after the injection commenced. Injection into the vessel from RCIC lasted four minutes and 23 seconds. Injection into the vessel from HPCS lasted 40 seconds.

The Reactor Water Cleanup System isolation, although not a Technical Specification requirement, utilizes the Nuclear Steam Supply Shutoff System (NS**4) logic to signal the closure of valves 2WCS*MOV102 and 2WCS*MOV12. The NS**4 logic system is an engineered safety system.

The other ESF actuations (GTS initiation, Primary Containment isolations Groups 2, 3, 4, 8 and 91, and Control Building Special Filter Train initiation) were also conservative plant responses with minimal plant impact and no resultant impact on public safety.

To satisfy the reportability requirements of Technical Specification 3.5.1.f, the following information is being provided for the ECCS High Pressure Coolant Injection event:

Total accumulated initiation cycles for the HPCS System (from receipt of the NMP2 operating license up to and including the December 12, 1991 event) = 5.

The usage factor value for the HPCS injection nozzle (as of December 12, 1991) remains significantly below 0.70.

IV. CORRECTIVE ACTION

Immediate operator actions involved restoring reactor vessel water inventory and commencing a controlled plant shutdown. Further, all automatic responses to RPV water Levels 2, 3, and 4 were verified to have properly occurred.

Additional corrective actions include:

1. Senior Reactor Operators (SROs) involved on shift have been removed from shift responsibilities until completion of a remediation program. Further disciplinary action is under consideration.

IV. CORRECTIVE ACTION (cont.)

2. Revising Operating Procedure N2-OP-3, "Condensate and Feedwater System" to require a 3-2-1 or a 3-3-1 Condensate, Condensate Booster and Feedwater operating pump alignment prior to starting a second Feedwater pump at high power operations.

3. Condensate pump 2CNM-P1C troubleshooting has been completed relative to the failure to start. No problems were found in the control circuit, fuses or control switch. The pump was able to be restarted satisfactorily following these checks.

4. A formal root cause for the WCS isolation is being performed. Upon completion of this investigation, an Engineering evaluation will be pursued regarding the possibility of defeating the WCS isolation signal (due to system sensitivity) during manual system operation with the system hot. If the root cause investigation identifies additional/alternate causes for the WCS isolation, a supplement to this LER will be submitted.

V. ADDITIONAL INFORMATION

A. Failed components: none.

B. Previous similar events:

The plant has experienced previous reactor scrams due to a loss of Feedwater flow resulting in an RPV water Level 3 trip signal. However, these events were not a result of a Condensate/Feedwater System transient induced by FWS/CNM pump misalignment. Therefore, corrective actions taken as a result of these previous events would not have helped prevent this scram from occurring. All scram event details are provided in Reactor Engineering Procedure N2-REP-6, "Post-Scram Review."

Additionally, the plant has experienced numerous Reactor Water Cleanup System isolations due to high differential flow trip signals. Preventative/corrective actions taken as a result of these events have been implemented for those specific situations addressed. Prior to September 25, 1989 (event date for Licensee Event Report 89-33), Operating Procedure N2-OP-37, "Reactor Water Cleanup System," provided operators the opportunity to defeat the delta flow isolation signal timer prior to system manipulation. Corrective actions taken as a result of this LER removed this step from the proced
re, requiring timers to be on during all system operations. Niagara Mohawk is presently re-evaluating this position via the WCS isolation root cause.

V. ADDITIONAL INFORMATION (cont.)

C. Identification of components referred to in this LER:

IEEE 803 EIIS IEEE 805
COMPONENT FUNCTION SYSTEM ID

Condensate System N/A SD
Feedwater System N/A SJ
High Pressure Core Spray System N/A BG
Reactor Core Isolation Cooling System N/A BN
Reactor Recirculation System N/A AD
Reactor Water Cleanup System N/A CE
Standby Gas Treatment System N/A BH
Residual Heat Removal System N/A BO
Primary Containment N/A NH
Secondary Containment N/A NG
Pump P SD/SJ
Reactor Vessel RPV SB
Injection Nozzle NZL BG

Main Turbine Generator TG TA
Emergency Diesel Generator DG EK
Switchgear SWGR EB

ATTACHMENT 1 TO 9201210249 PAGE 1 OF 1

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Vice President
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January 13, 1992

United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

RE: Docket No. 50-410
LER 91-23

Gentlemen:

In accordance with 10CFR50.73, we hereby submit the following Licensee Event Report:

LER 91-23 Is being submitted in accordance with 10CFR50.73 (a)(2)(iv), "Any event or condition that results in manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS)."

A 10CFR50.72 (b)(2)(ii) report was made at 0345 hours on December 12, 1992.

This report was completed in the format designated in NUREG-1022, Supplement 2, dated September 1985.

Very truly yours,

Joseph F. Firlit
Vice President - Nuclear Generation
JFF/GB/lmc
ATTACHMENT

xc: Thomas T. Martin, Regional Administrator Region I
Wayne L. Schmidt, Sr. Resident Inspector

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